



BRUCE H HAMILTON
Vice President
Oconee Nuclear Station

Duke Energy Corporation
ON01VP / 7800 Rochester Highway
Seneca, SC 29672

864 885 3487
864 885 4208 fax
bhhamilton@duke-energy.com

June 12, 2005

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-270
Licensee Event Report 270/2006-01, Revision 0
Problem Investigation Process No.: O-06-2079

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 270/2006-01, Revision 0, regarding a Unit 2 reactor trip following a trip of a reactor coolant pump during a maintenance activity.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A).

This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Bruce H. Hamilton, Vice President
Oconee Nuclear Site

Attachment

Document Control Desk

Date: June 12, 2006

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cc: Mr. William D. Travers
Administrator, Region II
U.S. Nuclear Regulatory Commission
61 Forsyth Street, S. W., Suite 23T85
Atlanta, GA 30303

Mr. L. N. Olshan
Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. M. C. Shannon
NRC Senior Resident Inspector
Oconee Nuclear Station

Mr. D. W. Rich
NRC Senior Resident Inspector
Oconee Nuclear Station

INPO (via E-mail)

Date: June 12, 2006

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bxc:

ONS Site:

bxc: ONS Site:

Document Control (Master File)*	PIP FILE*
Site PORC Members	
RGC MGR/B.G. Davenport	
RGC: Commitment Index/J.E. Smith#	LER Book*#
Vis Cntr Mgr/D.M.Stewart	
OPS-Procedures/D.B. Coyle#	
Work Control:D.V. Deatherage#	
Site Engineering:	
W.B. Edge#	T.A. Ledford#
K.R. Alter#	R.J. Freudenberger#
M. Bailey#	
EPIX Cord/B.R. Loftis.	

GO:

NR1A/R.L. Gill *	M.S. Tuckman#
ELL/EC050*	NSRB/E.B. Kulesa/EC05N*
NGO/SAA:D.J. Herrick	C.M. Misenheimer#
NGO/SA/S.B. Thomas	
NGO Serv: R.G. Hull#	
LEGAL/L.F. Vaughn*	RATES/M.J. Brown#

CNS:

SA MGR/T.D. Ray

RGC MGR/R. D. Hart

MNS:

SA MGR/J.R. Ferguson, Jr.

RGC MGR/C.J. Thomas

OPS Mgr/S.L. Bradshaw#

Non-routine Recipients:

Jeff Rowell

* - Hardcopy - All others by E-Mail Distribution

- Copied By Request: - All others by Directive

(Revised 1-17-2006)

LICENSEE EVENT REPORT (LER)

(See reverse for required number of
digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 50 hours.
Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME Oconee Nuclear Station, Unit 2				2. DOCKET NUMBER 05000- 0270				3. PAGE 1 OF 7			
4. TITLE Loss of Isolation during Pump Instrument Check Results in Reactor Trip											
5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED		
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER	
04	12	2006	2006	001	00	06	12	06	None	05000	
9. OPERATING MODE											
11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)											
10. POWER LEVEL											
100											
20.2201(b)											
20.2201(d)											
20.2203(a)(1)											
20.2203(a)(2)(i)											
20.2203(a)(2)(ii)											
20.2203(a)(2)(iii)											
20.2203(a)(2)(iv)											
20.2203(a)(2)(v)											
20.2203(a)(2)(vi)											
20.2203(a)(3)(i)											
20.2203(a)(3)(ii)											
20.2203(a)(4)											
50.36(c)(1)(i)(A)											
50.36(c)(1)(ii)(A)											
50.36(c)(2)											
50.46(a)(3)(ii)											
50.73(a)(2)(i)(A)											
50.73(a)(2)(i)(B)											
50.73(a)(2)(i)(C)											
50.73(a)(2)(ii)(A)											
50.73(a)(2)(ii)(B)											
50.73(a)(2)(iii)											
50.73(a)(2)(iv)(A)											
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50.73(a)(2)(v)(B)											
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50.73(a)(2)(v)(D)											
50.73(a)(2)(vii)											
50.73(a)(2)(viii)(A)											
50.73(a)(2)(viii)(B)											
50.73(a)(2)(ix)(A)											
50.73(a)(2)(x)											
73.71(a)(4)											
73.71(a)(5)											
OTHER											
Specify in Abstract below or in NRC Form 366A											
12. LICENSEE CONTACT FOR THIS LER											
FACILITY NAME B.G. Davenport, Regulatory Compliance Manager						TELEPHONE NUMBER (Include Area Code) (864) 885-3044					
13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT											
CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX		
14. SUPPLEMENTAL REPORT EXPECTED											
15. EXPECTED SUBMISSION DATE											
YES (If yes, complete EXPECTED SUBMISSION DATE)											
X NO											
16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)											
<p>On April 12, 2006, while operating at 100% in Mode 1, Oconee Nuclear Station Unit 2 experienced a reactor trip. The post trip response was normal, with all major operating parameters remaining within expected limits. Operators took appropriate action to stabilize the unit in Mode 3.</p> <p>The cause of the event was determined to be a trip of a reactor coolant pump during testing of a pump power transducer. The pump trip occurred due to a loss of isolation between the power transducer being tested and the operating circuit while connected to the test equipment. Corrective actions address changes to restrict use of this procedure on critical equipment while operating.</p> <p>This event is considered to have no significance with respect to the health and safety of the public.</p>											

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EVALUATION:

BACKGROUND

This event is reportable per 10CFR 50.73(a)(2)(iv)(A) as an event which resulted in a Reactor Trip at Oconee Nuclear Station.

As described in Section 7.2.2 of the Updated Final Safety Analysis Report (UFSAR), the Reactor Protective System (RPS) [EIIS:JC] has four redundant channels and uses 2 of 4 logic to initiate a reactor trip when two channels actuate. Each RPS channel has multiple functions which will cause the channel to trip when the applicable setpoint is reached. One such function is Flux/Flow/Imbalance. Flux is neutron flux, which indicates Reactor power. Flow is Reactor Coolant System (RCS) [EIIS:AB] flow. Imbalance is reactor power imbalance, defined as power in the top half of the core minus the power in the bottom half of the core, expressed as a percentage of full power. When the neutron flux signal exceeds a variable setpoint calculated from measured reactor coolant flow and reactor power imbalance, the associated protective channel trips.

The Integrated Control System (ICS) [EIIS:JA] provides automatic control to maintain the unit at the selected power level. In response to certain plant events, the ICS system will attempt to runback (automatically reduce power) to a predetermined setpoint. A Reactor Coolant Pump (RCP) [EIIS:P] trip is one of the events which initiates an ICS runback. However, this runback is not expected to occur fast enough to avoid a reactor trip if the RCP trip occurs at 100% power.

The Anticipated Transient Without Scram (ATWAS) [EIIS:JC] Mitigation System Actuation Circuitry (AMSAC) and the Diverse Scram System (DSS) provide non-safety back-up signals to mitigate an ATWAS event. DSS will trip the reactor in the event that the RPS fails to successfully trip the reactor and the transient results in a very high RCS pressure.

Prior to this event Unit 2 was operating at 100% power in Mode 1. Operations began activities to remove the Inadequate Core Cooling Monitor (ICCM) [EIIS:X1] train 2B from service for planned calibrations. The ICCM 2B train affects the operability of a number of safety systems, so the 2B trains of those systems were

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also logged out of service. However, both the Operations Control Room Senior Reactor Operator and the Instrument and Electrical (I&E) technicians involved in the planned work confirm that only the ATWAS systems (AMSAC and DSS) had physically been disabled (bypassed) at the time of the trip. As a result, at the time of the trip, no other safety systems or components were out of service that would have contributed to this event.

EVENT DESCRIPTION

At approximately 12:45 on April 12, 2006, a team of I&E technicians began work to inspect and test an AC watt transducer [EIIS:TD] for RCP 2B2 as part of a maintenance activity. They found no abnormal readings. By 13:35 hours they had completed their data collection task and were preparing to disconnect their test equipment.

At 13:35:51 RCP 2B2 tripped, which caused the ICS to initiate a runback. However, this runback is not expected to occur fast enough to avoid a reactor trip if the RCP trip occurs at 100% power.

At 13:35:57 RPS channels A and D tripped on Flux/Flow/Imbalance, which tripped the reactor.

Post trip response was normal. All control rod drive [EIIS:AA] breakers tripped and control rods dropped into the core. The turbine tripped as expected due to the reactor trip. Unit auxiliary power automatically transferred from the Normal to Start-up source as expected. Main Feedwater [EIIS:SJ] remained in service, with flow demand automatically reduced by the ICS. RCS temperature, RCS pressure, RCS inventory, Main Steam (MS) [EIIS:SB] Pressure, and Steam Generator (SG) [EIIS:SG] inventory remained within expected limits. No actuations or actuation demands occurred related to Emergency Feedwater [EIIS:BA] or Engineered Safeguards [EIIS:JE] (i.e. Emergency Core Cooling [EIIS:BG and BP], Containment Isolation [EIIS:NH], Containment Spray/Cooling [EIIS:BE and BK], and Emergency Power [EIIS:EK]). Since the RPS operated properly to trip the unit, the fact that AMSAC and DSS were out of service did not contribute to the event.

One control rod (Rod 7 Group 3) position indication showed the rod to be at 20% withdrawn while a second control rod (Rod 7 Group 2)

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position indication showed an out-limit light. The operators opened valves connecting the Borated Water Storage Tank to the suction of the operating High Pressure Injection (HPI) [EIIS:CB and BG] pump to increase the boron concentration in the RCS to ensure adequate shutdown margin. Further evaluation of other indications, including individual rod in-limit and group in-limit lights and core response, confirmed that both rods were fully inserted.

One Main Steam Relief Valve [EIIS:RV] (MSRV) (2MS-11, on the "B" header) did not reseal at 995 psig as expected. MS header pressure was lowered to 970 psig (89.8% of set pressure), at which point the valve reseated. The relief valve operation was reviewed by Valve Engineering personnel. Although the recorded reseal value (89.8%) was slightly lower than expected, it was not outside of the potential relief valve drift (3%) and blowdown (10%). Therefore, Valve Engineering concluded that the slight recorded variation of the reseal value (0.2% from the procedure value of 90%), was not indicative of a valve problem.

The post-trip investigation validated the plant response described above and confirmed the cause of the trip.

CAUSAL FACTORS

The Reactor Trip was a direct result of the trip of the 2B2 Reactor Coolant Pump. A root cause team was created to determine the cause of the RCP trip.

The Reactor Coolant Pump is powered from a three phase 6900 volt source. The pump protective relaying uses current transformer (CT) [ICT] circuits which allow control and monitoring devices to operate at reduced voltages with currents proportional to the current in the corresponding 6900 volt phase. One set, using 2 phases, of these CT circuits includes watt transducers which provide computer indication of the power used by the pump. This CT circuit also contains an electronic device which provides several protective relaying functions, including a phase balance current relay.

In this CT circuit only two phases, X and Z, are monitored and the third phase is calculated from the two monitored phases. The phase balance current relay function compares the measured and calculated

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currents. If an imbalance is detected between any two phases, real or calculated, the relay will trip the pump.

The work activity in progress involved connection of test instruments to one of the power transducers providing computer indication. The root cause team found that, while the test equipment was in place, there was a loss of isolation between the power transducer being tested and the operating CT circuit. The test setup involves isolation of the CT circuit by inserting a test plug into a knife switch. It is indeterminate if the loss of isolation occurred due to an interaction between the test plug and the knife switch or due to a problem with the knife switch itself.

The loss of isolation caused a portion of the current normally in the neutral circuit of the CT circuit to interact with the current injected by the test equipment and pass through ground as a return path, reducing the current in the neutral circuit. When the current in the neutral leg dropped, the protective relay detected the current unbalance and provided a trip signal to the pump breaker.

CORRECTIVE ACTIONS

Immediate:

1. Operators took appropriate actions to bring the unit to stable hot shutdown (Mode 3).

Subsequent:

1. The instrument procedure in use during this activity was placed on technical hold (which prevents its use) to address risk issues associated with performing this procedure on vital equipment at power.

Planned:

1. Additional troubleshooting will be performed during the next Unit 2 refueling outage to investigate potential failure modes involving the knife switch/test plug interactions.

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This corrective action is not considered to be an NRC Commitment item. There are no NRC Commitment items contained in this LER.

SAFETY ANALYSIS

This event did not include a Safety System Functional Failure.

A reactor trip is an anticipated transient and is considered the safe end state following many plant transients and most accident scenarios addressed in Chapter 15 of the UFSAR.

As described in Section 7.2.2 of the UFSAR, the flux/flow/imbalance parameter is one of the inputs which initiates a reactor trip. During this event, the unit tripped in response to actual plant parameters. All required systems and equipment operated as designed to mitigate the consequences of the RCP trip and stabilize the unit in Mode 3.

The minor discrepancies observed (two control rod position indications conflicting with the rod fully inserted indications, and the slight shift in reseal pressure of 2MS-11) had no adverse impact on the event.

Core Damage Significance (CDS) Impact

The CDS of this event has been evaluated quantitatively by considering the following:

- A reactor trip initiating event (reactor trip with no complications)
- Actual plant configuration and maintenance activities at the time of the trip
- Current Probabilistic Risk Assessment (PRA) model accounting for any necessary updates since the last revision

The event was an uncomplicated reactor scram with no appreciable impact on any safety systems.

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Large Early Release Frequency (LERF) Significance Impact

The LERF significance of this event has been evaluated quantitatively using the same considerations as for the CDS Impact. The event has an insignificant impact on the core damage risk.

Therefore, there was no actual impact on the health and safety of the public due to this event.

ADDITIONAL INFORMATION

A review of Unit trip events over the last five years indicates that there have been no similar reactor trip events at Oconee within that time period.

There were no releases of radioactive materials, radiation exposures or personnel injuries associated with this event.

The trip of the reactor constitutes a Maintenance Rule functional failure and is considered reportable under the Equipment Performance and Information Exchange (EPIX) program.

ENCLOSURE 1

SIGNATURE SHEET

PIP 0-06-2079

Prepared By: _____ Date: _____

Revised By: _____ Date: _____

Reviewed By: _____ Date: _____

_____ Date: _____

_____ Date: _____

_____ Date: _____

_____ Date: _____

_____ Date: _____

Approved By: _____ Date: _____

 Manager, RGC

Reviewed By: _____ Date: _____

 Station Manager

ENCLOSURES:

1. Regulatory Compliance Signature Sheet
2. References
3. Corrective Action Schedule
4. Personnel Contacted
5. Cause Code Summary

ENCLOSURE 2

REFERENCES

PIP O-06-2079

PIP O-06-2079

Root Cause Report

Post Trip Review Procedure (PT/O/A/0811/002) Trip 100 (completed procedure approved 4-24-06)

E-mail Dated 05/30/2006 From Sarah L Chisholm (PRA Group) with LER input from PRA Group

UFSAR Section 7.2.2

OEE-217-10 (Rev 010, per NEDL)

OSS-254.00-00-1033 Rev 19 RCS DBD (Section 5.3.1.4 on Reactor Coolant Pump 2B2)

ENCLOSURE 3

CORRECTIVE ACTION SCHEDULE

PIP 0-06-2079

<u>CORRECTIVE ACTION</u>	<u>PERSON(S) CONTACTED</u>	<u>PERSON(S) ASSIGNED TO</u>	<u>DUE DATE</u>
1	(From Root Cause Report)		

ENCLOSURE 4

PERSONNEL CONTACTED

PIP 0-06-2079

Ray Smith (OPS)
Jeff Rowell (RES)
Chad Hitt (RES)
Jim Kiser (MCE Valves)
Bill Rostron (RES)
Preston Gillespie (RES)
Eddie Anderson (OPS)
Sarah Chisholm (PRA Group)
Dennis Smith (I&E)

ENCLOSURE 5

CAUSE CODE ASSIGNMENT SHEET

PIP 0-06-2079

CAUSE CODES:

M2o Unanticipated Interaction Between Systems or Components